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Subject: ACCEPTANCE OF THE FEDERAL REPUBLIC OF GERMANY ISOTOPIC SOURCES OF RADIATION AND HEAT

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Fluor Daniel Hanford, Inc.
P.O. Box 1000
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August 7, 1997

FDH-9756992

Mr. T. K. Teynor, Director
Waste Programs Division
U.S. Department of Energy
Richland Operations Office
Post Office Box 500
Richland, Washington 99352

Dear Mr. Teynor:

ACCEPTANCE OF THE FEDERAL REPUBLIC OF GERMANY ISOTOPIC SOURCES OF RADIATION AND HEAT

References: Letter, J. M. Henning, RL, to President, WHC, "Receipt of Remote Handled Transuranic Waste (RH-TRU) in Solid Waste Facilities," 95-SWT-160/WPD:RFG, dated March 9, 1995.

Per Ms. Henning's letter (Attachment 1) Solid Waste Disposal is allowed to accept remote-handled transuranic (RH-TRU) waste if it is acceptable for final disposal at the Waste Isolation Pilot Plant (WIPP), or it has been appropriately addressed as part of a national program.

The Central Waste Complex (CWC) is preparing to receive eight RH-TRU casks containing 34 borosilicate glass logs from the 324 Building, B&W Hanford Company (BWHC). As packaged, this waste does not meet WIPP acceptance criteria nor has it been addressed as part of a national program for disposal. In addition, it has been determined that the source material for two of the glass logs, Federal Republic of Germany (FRG) A and B, originated from a commercial source. However based on process knowledge, they do meet the definition of RH-TRU. Information supporting the RH-TRU designation prepared by Waste Management Federal Services of Hanford, Inc. (WMH) is attached (Attachment 2). Additionally, the draft Waste Certification Summary prepared by BWHC is attached (Attachment 3).

WMH proposes to manage the FRG Isotopic Heat and Radiation Sources as RH-TRU waste at the CWC until a final disposal pathway is identified. The storage of this waste at CWC is within the Safety Assessment submitted to RL for approval.

My staff has reviewed the RH-TRU designation and Waste Certification Summary and concur in those determinations. The storage of this waste at the CWC until a final disposal pathway is identified is in the best interest of the site.

The Sr was obtained from residual strontium nitrate solutions contained in piping and tanks when B Plant was shut down. These solutions were purified using a carbonate precipitation process to achieve the desired composition for the USDOE/FRG project. A major objective of this purification was to reduce the sodium content that would be detrimental to ultimate glass leaching resistance. These solutions were not subjected to the same purification steps required to produce isotopic capsules; therefore, some residual uranium and plutonium "impurities" remained.

Both the Cs and Sr were transferred from B Plant to the 324 Building (operated by Pacific Northwest Laboratories [PNL]) as nitrate solutions.

PNL incorporated the Cs and Sr into the glass by adding the liquid nitrate solutions directly into the glass batch using the Radioactive Liquid Feed Ceramic Melter in the 324 Building Cell B.[3] Following homogenization in the melter, PNL poured the glass into metal canisters.

Material that began as spent fuel reprocessing waste, i.e., the highly radioactive Cs and Sr components of the liquid waste, was converted into product at B Plant. That product was incorporated into the borosilicate glass canisters fabricated within the 324 Building. Only after the USDOE/FRG project was terminated and no other beneficial use could be found for the canisters, did the material revert to waste.

RH-TRU Designation Support Information:

The Nuclear Regulatory Commission defines High Level Waste (HLW) in 10 CFR 60 as (1) irradiated reactor fuel, (2) liquid wastes resulting from the operation of the first cycle solvent extraction system, or equivalent, and the concentrated wastes from subsequent extraction cycles, or equivalent, in a facility for reprocessing irradiated reactor fuel, and (3) solids into which such liquid wastes have been converted.

Using the definition above, the thirty-two German Logs do not meet the definition of HLW for the following reasons:

1) The material was not irradiated reactor fuel. The material was liquid waste resulting from the reprocessing of spent N Reactor fuel at PUREX.

2) The material was not liquid waste resulting from the operation of the first cycle extraction system or any extraction cycle in a facility for reprocessing irradiated reactor fuel. These materials were considered product at the time of extraction. In addition, B-Plant was never used to reprocess irradiated reactor fuel.

3) The cesium and strontium recovered during the PUREX process were not waste materials at the time of recovery. The isotopes were recovered and used as feed for scientific research to later be used as a product (isotopic heat sources).

Using the definition above, the two prototype canisters, FRG A and B, also do not meet the definition of HLW for the following reasons:

Background Information to Support the Babcock & Wilcox RH-TRU Designation of the FRG Isotopic Heat and Radiation Sources

Process Knowledge of Source Material

A. Prototype Canisters - FRG A and FRG B

The source material for the radionuclides in the FRG A and B canisters was spent fuel discharged from the Point Beach reactor in Two Rivers, WI in May 1974 (burn up was 29600 Mwd/tU). Originally, this fuel was obtained by DOE for use in the Nuclear Waste Vitrification Project (Contract EY-76-C-06-1830), the objective of which was to demonstrate the vitrification of actual high-level liquid waste from spent LWR fuel. A portion of this fission product stream was used for the FRG project.

After acquisition from West Valley, NY, the spent fuel underwent chop/leach and dissolution in the 324 Building and was then sent to the nearby 325 Building for separation. Following separation, the fission product stream was returned to the 324 Building and incorporated into a borosilicate glass to demonstrate the vitrification process. Residual fission product stream from this project was subsequently used to produce the prototypes for the FRG project.

A "spiked" glass was necessary to check out and demonstrate the new Radioactive Liquid Feed Ceramic Melter (RLFCM) and the rest of the equipment designed for the FRG project. Thus, this glass was product insofar as it was required to develop the RLFCM and confirm its operation, and confirm the capability to fabricate and characterize the FRG canisters. Following checkout of the melter, the glass was poured into canisters A and B and further tests were conducted to verify the FRG canister welding process and the canister characterization equipment. Thus, the canisters themselves were "product".

B. German Logs: FRG Isotopic Heat Sources - 32 Canisters

The ^{137}Cs and ^{90}Sr incorporated into the borosilicate glass were extracted from the liquid waste resulting from reprocessing spent N Reactor fuel at PUREX. Following transfer of the reprocessing liquid waste to Tank Farm storage, and subsequent transfer to B Plant, a crude separation using ion exchange was performed on the supernate to obtain highly radioactive Cs and Sr. The Sr came from this source following modest additional purification; more extensive purification was performed to yield the Cs. The purification activities that produced the FRG Cs and Sr feed materials were performed in the mid-1980s.

Cs purification following initial separation from the reprocessing supernate was accomplished by centrifuging to remove solids, followed by additional ion exchange separation to achieve final purification.(1) Following conversion of the nitrate to chloride and precipitation, the CsCl was encapsulated to produce isotopic sources. The Cs used for the fabrication of the borosilicate glass was obtained by cutting open 187 CsCl capsules and converting the chloride back to a nitrate solution. A list of these CsCl capsules is provided in the Waste Certification Summary.

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ATTACHMENT 2

"Background Information to Support the BWHC RH-TRU

Designation of the FRG Isotopic Heat and Radiation Sources"

Consisting of 4 pages,

Including cover page

Department of Energy
Richland Operations Office
P.O. Box 550
Richland, Washington 99352

9501341

MAR 09 1995

95-SWT-160

President
Westinghouse Hanford Company
Richland, Washington

Dear Sir:

RECEIPT OF REMOTE HANDLED TRANSURANIC WASTE (RH-TRU) IN SOLID WASTE FACILITIES

Reference: Letter, J. M. Hennig to President, WHC, dated October 6, 1994, same subject as above (94-SWT-609).

The referenced letter requested WHC Solid Waste Division (SWD) to accept no RH-TRU derived from fuel, other than waste designated for storage in project W-272, until further clarification was received from HQ. Attached is a memorandum further clarifying the definition of RH-TRU to be used at Hanford. SWD is requested to reinitiate RH-TRU receipts, subject to the requirements of safety analysis requirements and the certification requirements of WHC-EP-0063-4. Only RH-TRU is planned to be transferred to SWD. The referenced letter is hereby cancelled.

Please direct any questions that you may have on this subject to R. F. Guercia, Program Manager for Solid Waste, on 376-5494 or R. M. Gordon, of my staff, on 372-2139.

Sincerely,

June M. Hennig, Director
Waste Programs Division

WPD:RFG

Attachment

cc w/attach:
J. G. Riddelle
R. D. Pierce,
P. L. Hapke, W

FDH-9756992

ATTACHMENT 1

"Receipt of Remote Handled Transuranic Waste (RH-TRU)

In Solid Waste Facilities"

Consisting of 2 pages,

Including cover page

I have directed WMH to proceed with their proposed course of action. If you have any questions please contact Mr. R. L. Bisping of my staff at 376-5695 or Mr. G. C. Triner, of WMH, at 372-0771.

Sincerely,

C. G. Mattsson, Director
Waste Management

CGM/rlb/dls

Attachment (3)

FDH-9756992

ATTACHMENT 3

"B&W Hanford Company Waste

Certification Summary"

(Excluding Appendices)

Consisting of 28 pages,

Including cover page

1) Although the source material originated from irradiated reactor fuel, it was not waste but product originally targeted to the DOE-funded Nuclear Waste Vitrification Project with the residual fission product stream used to fabricate the prototype canisters for the FRG program.

2) Although the source material was from the first cycle extraction at the 325 Building, it was product rather than waste.

3) The fission products were product rather than waste. They were captured to demonstrate the vitrification of fission products in the NWVP and FRG programs.

DOE order 5820.2a defines transuranic (TRU) waste as "Without regard to source or form, waste that is contaminated with alpha-emitting transuranic radionuclides with half-lives greater than 20 years and concentrations greater that 100 nCi/g at the time of assay".

Per P. J. Weaver, samples of four canisters were taken and analyzed for plutonium and showed levels ranging from 690-1140 nCi/g. Plutonium levels for the remaining 26 canisters were calculated to range between 437-1056 nCi/g using the four sampled canisters as a reference. Calculation of the TRU content of FRG A (the canister with the least Pu content) results in 120 nCi/g. These levels of plutonium combined with high dose rates would require the canisters to be designated as remote handled (RH) TRU.

Additionally, the Carlsbad Area Office Interim Guidance on Ensuring That Waste Qualifies for Disposal at the Waste Isolation Pilot Plant can also be used to support the RH-TRU designation.

Using the TRU Waste Decision Tree, the RH-TRU designation is supported.

1. Is it material withdrawn from nuclear reactor after irradiation? No, it is irradiated reactor material that has been processed to produce a product.

2. Waste from first cycle separation? No, because the material was "product" rather than "waste".

3. Go to CH and RH TRU decision tree.

4. Radionuclides with atomic number > 92, half-life > 20 years, alpha emitter > 100 nCi/g of waste? Yes.

5. Surface does rate greater than 200 mrem/hour? Yes.

Per the decision tree, the waste is RH-TRU.

B&W HANFORD COMPANY
WASTE CERTIFICATION SUMMARY

Effective Date:
Summary #:
WSRD #: 20A-00

Waste Form: Remote-Handled Transuranic Waste Canisters

1.0 WASTE GENERATING PROCESS DESCRIPTION

1.1 Background

In November 1984, the U.S. Department of Energy (USDOE) and the Federal Minister for Research and Technology of the Federal Republic (BMFT) of Germany signed a project agreement to develop processes for the treatment and immobilization of high-level radioactive waste and to fabricate isotopic sources of radiation and heat. The isotopic sources of heat and radiation that the USDOE fabricated were to be used as part of a repository testing program in the Asse Salt Mine in Germany. The Project Agreement was amended in June 1985 to specify fabrication of 30 isotopic heat sources in three sets of 10 sources each for shipment to Germany. In March and April 1987, the Project Agreement was again amended to include fabrication of another two instrumented isotopic heat sources with an average heat generation rate of 20 W per liter for use in obtaining data for a heat transfer validation model. In 1992, the BMFT was compelled to cancel the project because it could not obtain formal licensing for conducting its experiments at Asse. Some time later (i.e., in December 1995), DOE and the BMFT signed a closeout agreement for the project. BMFT agreed to furnish six cast iron casks (CASTOR GSF-5) and two stainless steel (GNS-12) casks for DOE use in transporting and storing the canisters; DOE and BMFT agreed that BMFT was not the proprietor of the canisters.

PNL fabricated two other canisters to test the fabrication equipment before they fabricated the 30 isotopic heat sources and the two instrumented canisters. The two test canisters (FRG A and B) are described in the Process Description section below.

On January 14, 1997, the B&W Hanford Company (BWHC) Project Manager for Project C-229 designated the 34 borosilicate glass filled canisters that were related to the USDOE/BMFT project as non-hazardous, remote-handled, transuranic waste (RH-TRU).

1.2 Process Description

1.2.1 Source Material

The ^{137}Cs and ^{90}Sr incorporated into the borosilicate glass were extracted from the liquid waste resulting from reprocessing spent N Reactor fuel at PUREX. Following transfer of the reprocessing liquid waste to Tank Farm storage, and subsequent transfer to B Plant, a crude separation using ion exchange was performed on the supernate to obtain highly radioactive Cs and Sr. The Sr came from this source following modest additional purification; more extensive purification was performed to yield the Cs. The purification activities that produced the FRG Cs and Sr feed materials were performed in the mid-1980's.

Cs purification following initial separation from the reprocessing supernate was accomplished by centrifuging to remove solids, followed by additional ion exchange separation to achieve final purification.[1] Following conversion of the nitrate to chloride and precipitation, the CsCl was encapsulated to produce isotopic sources. The Cs used for the fabrication of the borosilicate glass was obtained by cutting open 187 CsCl capsules and converting the chloride back to a nitrate solution. A list of these CsCl capsules is provided in Appendix 1.

The Sr was obtained from residual strontium nitrate solutions contained in piping and tanks when B Plant was shut down. These solutions were purified using a carbonate precipitation process[2] to achieve the desired composition for the USDOE/FRG project. A major objective of this purification was to reduce the sodium content that would be detrimental to ultimate glass leaching resistance. These solutions were not subjected to the same purification steps required to produce isotopic capsules; therefore, some residual uranium and plutonium "impurities" remained. Based on the ^{240}Pu content of the residual plutonium (8.17 wt% ^{240}Pu in plutonium), the source material in all the glass logs except FRG A and B originated primarily from defense-related production activities. The source material for FRG A and B was derived from commercial power reactor spent fuel (see FRG A and B Radiological Characterization).

Both the Cs and Sr were transferred from B Plant to the 324 Building (operated by Pacific Northwest Laboratories [PNL]) as nitrate solutions.

PNL incorporated the Cs and Sr into the glass by adding the liquid nitrate solutions directly into the glass batch using the Radioactive Liquid Feed Ceramic Melter in the 324 Building Cell B.[3] Following homogenization in the melter, PNL poured the glass into metal canisters.

Material that began as spent fuel reprocessing waste, i.e., the highly radioactive Cs and Sr components of the liquid waste, was converted into product at B Plant. That product was incorporated into the borosilicate glass canisters fabricated within the 324 Building. Only after the USDOE/FRG project was terminated and no other beneficial use could be found for the canisters, did the material revert to waste.

1.2.2 Canister Fabrication

Under contract to the U.S. Department of Energy, Pacific Northwest Laboratories (PNL) filled a total of 34 stainless steel canisters with a borosilicate glass matrix that contained various quantities of radioisotopes. PNL used the remotely operated radioactive liquid-fed ceramic melter (RLFCM) in the 324 Building Radiological Engineering Cell B to fill the canisters.

PNL created three types of canisters. First, they fabricated two test canisters (designated FRG A and B) to test the glass fabrication process during the RLFCM-6 run in November 1985. Then PNL fabricated 30 canisters in three batches (RLFCM Runs 7, 8, and 9) of 10 canisters each to meet the criteria established in the DOE/BMFT agreement. Finally, PNL fabricated two more instrumented canisters for BMFT (Canisters 13 and 19) after the 30 canisters were fabricated. Those two canisters were to be used by BMFT to study the effects of heat on the borosilicate glass. Table 1 contains a chronology of the fabrication activities.

Table 1. Chronology of Canister Fabrication

Fabrication Dates	Fabrication Campaign Designation	Total RH-TRU Waste Canisters Fabricated	RH-TRU Waste Canister Numbers
11/19/85 - 11/20/85	RLFCM-6	2	A, B
2/1/86 - 10/23/86	RLFCM-7	10	1, 2, 3, 5, 6, 7, 8, 12, 17, 14
12/17/86 - 2/1/87	RLFCM-8	10	21, 28, 34,33, 36, 37, 41, 42, 43, 44
3/6/87 - 3/22/87	RLFCM-9	10	10, 18, 20, 38, 45, 46, 47, 48, 49, 50
3/27/87 - 3/29/87	RLFCM-10	2	13, 19

To fabricate the canisters, PNL blended radioactive feed slurries with glass forming chemicals then calcined the liquid waste and melted it to form a borosilicate glass. The molten glass was airlifted into stainless steel canisters. The 30 canisters fabricated in RLFCM runs 7, 8, and 9 were filled in an average of three glass pours per canister. One test canister (FRG A) was filled using two pours; the other test canister (FRG B) was filled in four pours. The two instrumented canisters (13 and 19) received glass in two pours each.

After the 30 canisters were covered, they were allowed to cool, then stored in B Cell for approximately one year. PNL retrieved each canister in order from storage, cleaned the weld surface, inserted a calibrated helium capsule in the void space, then welded a lid onto the canister using an autogenous gas tungsten arc (GTA) welding process. PNL conducted a gross leak check and visual weld examination of each welded canister, rinsed each canister with water to remove loose contamination, then passed it into an air lock between Cell B and Cell A. In the airlock, PNL verified the integrity of each weld using a helium leak detection system and measured the canister surface-exposure rate. When the leak check was complete, PNL transferred each canister to Cell A, where it was decontaminated using an electro-polishing process. PNL then placed each electro-polished canister in a water-cooled storage rack.

PNL welded lids on each of the four other canisters in the same manner as the 30 canisters. Test canisters FRG A and B were not leak checked because they were not fabricated to meet the leak criteria specified in the Project Agreement. Canister FRG A was not electro-polished because it was to be

stored on the floor of Cell A, where the levels of non-fixed contamination were judged to be more than those on the canister.

Before the instrumented canisters (13 and 19) were welded shut, PNL removed external instrumentation attached to the canisters. After welding, the two instrumented were leak checked; canister 13 passed the sensitive leak check. Canister 19 could not be fine leak checked successfully because the pressure could not be reduced to less than 10 um, as required for the sensitive leak check. Tests of the canister showed that the weld around the canister lid did not leak. Subsequent tests conducted on gas samples from the test vacuum vessel showed that the canister leaked from two locations on the wall.

The two instrumented canisters (13 and 19) and the two test canisters (A and B) are stored in 324 Building Cell A with the 30 other TRU-RH waste canisters. The two test canisters are sitting on the cell floor next to cell window, and the two instrumented canisters are inside of the rinse tank system and the ovality measuring equipment.

The RH-TRU waste canisters packaged in the storage/transportation casks do not meet the Solid Waste Management Acceptance Criteria; therefore, final acceptance will be determined by DOE-RL Waste Programs Division. The *Hanford Site Solid Waste Acceptance Criteria* (WHC-EP-0063)[4] Section 5.0 TRANSURANIC WASTE lists requirements for TRU waste generators. The RH-TRU waste canisters will be contained in two forms of transportation/storage casks (see PACKAGING below), rather than 55-gal drums or standard waste boxes that can be used without evaluation that are described in Section 5.4.1.2. The casks were evaluated and found acceptable to transport the RH-TRU waste canisters (see PACKAGING below).

The waste package assemblies will exceed the weight limits for drums and boxes given in Section 5.4.1.2. Weight limits were established in the safety documentation for the casks to ensure safe handling and transport (see PACKAGING below); therefore, separate DOE-RL approvals to exceed the weight limits will not be required.

The surface dose rates of the packages will not exceed the 0.001 Sv/hr (100 mrem/hr) limits given in 5.4.2.2 (see the discussion in External Shielding below). The inclusion of the casks in the Central Waste Complex (CWC) was evaluated as part of the design of the storage area. The Interim Storage Area (ISA) where the casks will be stored will be fenced and locked after the casks are placed on the storage pad. The ISA is located away from areas subject to routine traffic, thereby ensuring reduced exposures to CWC personnel.

The casks containing the RH-TRU waste canisters will be stored outside, within the CWC boundaries, for approximately 20 years. A storage pad with a weather cover will be constructed to contain the loaded casks at the CWC. The pad will be surrounded by a fence to prevent access. Actions will be then taken after that time to move the canisters to a suitable repository for long-term storage. DOE-RL Waste Programs must approve the acceptance of the RH-TRU canisters because a final disposal site and a path forward for their disposition has not been established.

2.0 PHYSICAL DESCRIPTION

The RH-TRU waste canisters are comprised of 34 isotopic heat sources contained in a borosilicate glass matrix poured into stainless steel canisters measuring

1197 mm in height and 300 mm outside diameter, with 8 mm thick walls.[5] The average weight of an empty canister is 79.8 kg, and the average weight of a filled canister is 327 kg.[3] Table 2 contains the physical characteristics for each of the 34 RH-TRU waste canisters. PNL determined the volume of the glass pours for the canisters (RLFCM Runs, 7, 8, and 9) by using the RLFCM glass level detection system or melter dropout data.[3]

Table 2. Physical Description of RH-TRU Waste Canisters as Fabricated

RLFCM Run No.	RH-TRU Canister No.	Filled RH-TRU Canister Mass, (kg)	Glass Mass, (kg)	Glass Volume, (l)
6[a]	A	134.1	54.3	--
6[a]	B	230.9	151.1	--
RLFCM-7[b]	1	247.7	169.0	60.9
	2	245.4	166.7	59.3
	3	235.4	158.7	59.4
	5	234.1	155.8	59.8
	6	231.3	153.2	59.3
	7	225.9	148.5	63.5
	8	241.8	163.4	59.4
	12	249.5	171.4	63.1
	17	235.4	157.3	60.6
	14	235.0	156.4	60.6
RLFCM-8[b]	21	231.3	153.2	60.2
	28	243.1	165.7	64.2
	34	240.9	163.2	63.6
	33	222.3	143.6	61.8
	36	240.9	162.5	64.2
	37	229.5	151.4	58.6
	41	240.4	158.7	60.9
	42	235.9	155.5	60.2
	43	235.0	155.0	60.2
	44	231.8	152.3	58.6
RLFCM-9[b]	10	236.8	159.1	59.8
	18	237.2	159.5	60.2

RLFCM Run No.	RH-TRU Canister No.	Filled RH-TRU Canister Mass, (kg)	Glass Mass, (kg)	Glass Volume, (l)
	20	236.8	158.5	59.8
	38	238.1	159.5	59.3
	45	237.2	158.6	59.7
	46	235.0	156.8	59.1
	47	240.9	161.1	59.3
	48	243.6	164.7	62.9
	49	233.1	154.3	61.0
	50	236.3	155.3	59.8
RLFCM-10	13	229.5[c]	141.3[d]	56[e]
	19	229.1[f]	145.3[d]	53[g]

(a) Information from BNWL Log Book 52694
(b) PNL 6790, Appendix C
(c) Canister 13 FRG Canister Fabrication Run Book Rev. 0 - 2/88 - Page 136 of 157
(d) Letter, L. K. Holten to Dr. I. Muller-Lyda, see Appendix 2
(e) Canister No. 13 RLFCM Data Sheet No. 14 Page 63 of 65
(f) Canister 19 FRG Canister Fabrication Run Book Rev. 0 - 2/88 - Page 136 of 157
(g) Canister No. 19 RLFCM Data Sheet No. 14 Page 63 of 65

3.0 RADIOLOGICAL CHARACTERIZATION

PNL performed radiological characterization for each of the 34 RH-TRU waste canisters as it was fabricated.[3] The information provided below originated in PNL; it can be considered as conservative because it does not reflect the radiological decay that has occurred since fabrication. No canister contains more than 663 DE-Ci. Also, each canister contains less than 3.0 g fissile material.

3.1 FRG A and B Radiological Characterization

The source material for the glass in FRG A and B canisters was different from that used for the other 32 canisters. The radionuclide source material was left from the Nuclear Waste Vitrification Project (NWVP) that had just been completed in the 324 Building. The goal of the NWVP was to demonstrate the vitrification of high-level liquid waste from spent Light Water Reactor (LWR) fuel. As such, this material is not defense related but came from spent commercial reactor fuel. The NWVP residual used to spike the glass for RLFCM Run 6 began as spent fuel from the Peach Point reactors in Two Rivers, WI. PNL-3038[6] describes the creation of a small pilot plant to perform chop/leach in the 324 Bldg. B-Cell, separation in the 325 Bldg. using the basic PUREX flowsheet, and incorporation into glass back in the 324 Bldg. B-Cell to demonstrate the overall process.

RLFCM-6 was conducted to complete checkout of the RLFCM melter and the rest of the equipment and processes associated with the FRG project prior to committing the Cs and Sr sources designated for the FRG canisters.[7] The glass poured into canisters FRG A and B was "spiked" with the residual NWVP fission product stream material to verify operation of the melter liquid level indicators. The melter was essentially emptied in the process of pouring these two canisters.

FRG A and B contained very small quantities of plutonium, as well as cesium and uranium isotopes. The plutonium concentration in the FRG A and B glass was significantly less than what is normally expected from the PUREX flowsheet because of the laboratory aspect of the separation operation associated with the NWVP. Table 3 shows the target glass activity in the RLFCM-6 Run used to create the test canisters (FRG A and B). Table 3 also shows the calculated radionuclide content in FRG A and B, based upon the specific activity in the target glass and the calculated value of the glass mass. Plutonium values in the table in the RLFCM-6 Run Plan[7] and Westsik report[8] are suspect (relationships between target activities in feed material and the glass are consistent for all constituents except Pu); therefore, the plutonium content is based on the europium content of the glass. From PNL-3038,[6] Table 4.7, which defines the composition of the Point Beach Reactor fuel fission product stream after dissolution, the ratio of Pu to Eu is 40.6 g to 150.0 g = 0.2707. Multiplying this ratio by the europium content of the RLFCM-6 glass [obtained by dividing the europium activity by the europium specific activity (2.64E+02 Ci/g Eu)] yields a Pu glass content of 0.0226 g. A similar calculation for FRG B yields 0.0629 g plutonium. Because this plutonium came from LWR fuel which had undergone burnup of 29,500 MWD/tU, the ^{240}Pu concentration was 22.81 wt% in total plutonium. Calculation was required because no sampling data were available for the two test canisters.

Table 3. Radiochemical Activity of FRG A and B Canisters (Major Isotopes Only)

Isotope	Specific Activity in RLFCM-6 Glass Target Value, (Ci/Kg) [7,8]	FRG A Calculated Activity, Ci [a]	FRG B Calculated Activity, Ci [a]
^{134}Cs	0.55	2.99E+1	8.31E+1
^{137}Cs	81.14	4.41E+3	1.23E+4
^{137m}Ba	77.84	4.23E+3	1.18E+4
^{232}Th	4.53E-7	2.46E-5	6.86E-5
U (all isotopes)	--	--	--
^{154}Eu	0.40	2.17E+1	6.04E+1
Pu (all isotopes)	0.002	2.26E-2	6.29E-2

(a) Assumes canister shell mass = 79.8 kg, FRG A glass mass = 54.3 kg, FRG B glass mass = 151.1 kg

Table 4 contains mass levels of certain species present in the RLFCM-6 glass.

Table 4. Mass Levels of Species in Reference Glass for RLFCM-6 [7,8]

Species	Wt% in RLFCM-6 Glass
ThO_2	0.94
UO_2	0.15
PuO_2	negligible
CsO_2	0.11
PuO_2	0.04

3.2 RLFCM-7, 8, 9, and 10 Radiological Characterization

PNL filled the 30 canisters with borosilicate glass containing varying amounts of radionuclides, principally ^{137}Cs and ^{90}Sr. Trace quantities of plutonium, americium, and uranium isotopes also appeared in the glass. The ^{137}Cs in the glass originated from the conversion of CsCl capsules, and the ^{90}Sr originated from residual strontium nitrate solutions left over after tank cleaning and line flushing activities conducted when B Plant was shut down that were purified using a carbonate precipitation process. The major source of plutonium was the unpurified waste solutions from B Plant; a lesser source was the residual plutonium present in the melter from earlier tests. PNL did not anticipate that the waste solutions from B Plant would contain plutonium

because prior analyses showed the quantity of fissile material was below the normal detection limits.
Table 5 shows average properties of the 30 RH-TRU waste canisters fabricated during RLFCM Runs 7, 8, and 9, and the two instrumented RH-TRU waste canisters fabricated during RLFCM Run 10.[5]

Table 5. Average Radiochemical Characteristics of 32 RH-TRU Canisters as Fabricated

RLFCM Run	Average ^{137}Cs Content per Canister, kCi	Average ^{90}Sr Content per Canister, kCi	Average Decay Heat per Canister, watts	Average Surface Exposure Rate, R/hr
RLFCM-7	192	85	1490	272,000
RLFCM-8	78	143	1330	112,000
RLFCM-9	207	130	1860	310,000
RLFCM-10	136	59	1047	210,000

3.2.1 RLFCM-7, 8, and 9 Radiological Characterization

For the RLFCM-7 run, PNL's target concentration of cesium oxide was 5.73 wt% and strontium oxide was 1.9 wt%.[3] The strontium oxide was determined directly from ICP analysis of glass samples collected during pouring. The cesium oxide concentration was determined using radiochemical analytical results to back-calculate cesium oxide from known isotopic abundance of ^{137}Cs. Radiochemical analysis was conducted for the ^{90}Sr isotope.

In the RLFCM-8 run, PNL increased the concentration of strontium and decreased the concentration of cesium in the canisters from the RLFCM-7 run to produce 10 canisters with a constant decay heat. No cesium fraction was added to the melter feed slurry from that present at the end of RLFCM-7; therefore, cesium declined throughout the filling of the 10 canisters. To compensate for high concentrations of cesium present at the beginning of RLFCM-8, PNL added a feed with a high strontium content (low exposure, high decay heat), increasing the concentration of strontium through the production of the fifth canister (Canister 36). Errors in the sample analyses caused the ^{90}Sr concentrations to decrease slowly, rather than increasing at a more gradual rate as planned in the last five canisters in RLFCM-8. As a result, canisters 37, 41, 24, 43, and 44 had a decay heat lower than specified in the project agreement.

PNL produced 10 RH-TRU canisters with the highest decay heat and surface exposure rates during RLFCM-9. Concentrations of cesium and strontium were constant through the fifth canister (Canister 45). When supplies of strontium became insufficient, PNL substituted cesium in the glass for the last five canisters (46, 47, 48, 49, 50). The concentration of cesium increased at a higher rate than the concentration of strontium decreased. Table 6 below shows the disproportionate increase of cesium with respect to strontium in the last five canisters.

During the fabrication of the 30 RH-TRU waste canisters, PNL collected a sample of the glass from two of the three pours and analyzed ^{137}Cs using gamma spectrometry and ^{90}Sr using cation exchange separation and beta counting.[3] If PNL did not collect a glass sample, they estimated the composition of the glass pour from analyses of the preceding and subsequent pours. PNL used the concentration of the ^{137}Cs and the ^{90}Sr directly to determine the decay heat in each of the 30 RH-TRU canisters.[3]

Table 6 lists the ^{137}Cs and the ^{90}Sr contents (measured shortly after fabrication - current activity is less because of decay) of the 30 RH-TRU waste canisters.[3]

Table 6. Quantities of Cesium and Strontium in 30 RH-TRU Waste Canisters

Canister Number	Cesium, kCi	Strontium, kCi
1	190	100
2	196	110
3	175	99
5	193	88
6	198	75
7	198	73
8	237	88
12	197	77
17	178	67
14	190	70
21	187	90
28	170	130
34	137	147
33	55	146
36	80	158
37	47	156
41	38	159
42	26	146
43	19	150
44	17	145
10	182	148
18	197	143

Canister Number	Cesium, kCi	Strontium, kCi
20	194	142
38	198	143
45	191	137
46	216	138
47	212	126
48	233	120
49	221	103
50	221	105

3.2.1.1 Plutonium, Uranium, and Americium Estimations for 30 RH-TRU Waste Canisters

Characterization of plutonium and fissile material present in the 30 RH-TRU waste canisters is based upon PNL characterization activities (see Appendix 2). To determine the plutonium isotopes present, PNL collected glass samples from RH-TRU waste canisters 3, 10, 36, and 38. Using mass spectrometry, PNL analyzed for $^{239,240}Pu$ to ^{90}Sr ratios in the four samples and then estimated the quantities of plutonium in the remaining 26 canisters. Table 7 shows the correlation between the ^{90}Sr concentration and the $^{239,240}Pu$ concentration for these four RH-TRU waste canisters.

Table 7. Correlation of $^{239,240}Pu$ to ^{90}Sr in 30 RH-TRU Waste Canisters

Canister No.	^{90}Sr Concentration (Ci/g)	$^{239,240}Pu$ Concentration (uCi/g)	Ratio $^{239,240}Pu/^{90}Sr$
3	0.623	0.40	0.64
10	0.820	0.61	0.75
36	0.975	0.79	0.81
38	0.895	0.57	0.64

PNL estimated the quantities of ^{238}Pu, ^{241}Pu, and ^{242}Pu using mass percentages shown in Table 8. Table 8 data were obtained by averaging three separate analyses from two separate RH-TRU waste canisters (10 and 36) to calculate the isotopic mass data and then duplicating the analysis of the canister 10 glass sample to determine analytical variability. The variance in the three isotopes was much greater than that of ^{239}Pu and ^{240}Pu. Calculation of the plutonium in subsequent calculations was based on the $^{239,240}Pu$ content, rather than total plutonium content, to develop the correlation of plutonium to ^{90}Sr.

Table 8. Mass Percentages of Plutonium Isotopes in 30 RH-TRU Waste Canisters

Plutonium Isotope	Mass Percent (%)
238	0.34
239	90.00
240	8.17
241	0.77
242	0.39

PNL used separate statistical approaches to estimate the quantities of each of the five plutonium isotopes separately for the four canisters for which glass samples were analyzed, and then for the other 26 RH-TRU canisters. Appendix 2 contains correspondence from PNL to the BMFT that documented the fissile material content and the plutonium isotope content. Table 9 shows estimates of the five plutonium isotopes present in each of the 30 RH-TRU waste canisters as PNL estimated them in 1989 for the BMFT (Appendix2, Holten to Stipler letter, Table 4). These Pu values are conservative in that a margin is included to provide confidence that they bound the actual content.

Table 9. Plutonium Isotopes Present in 30 RH-TRU Waste Canisters

Canister No.	^{238}Pu/ Canister (Ci,g)	^{239}Pu/ Canister (Ci,g)	^{240}Pu/ Canister (Ci,g)	^{241}Pu/ Canister (Ci,g)	^{242}Pu/ Canister (Ci,g)
1	0.19 0.011	0.07 1.070	0.03 0.112	1.86 0.018	4.3E-5 0.011
2	0.21 0.012	0.08 1.155	0.03 0.122	2.12 0.020	4.7E-5 0.012
3	0.23 0.013	0.10 1.614	0.04 0.165	2.69 0.025	6.7E-5 0.017
5	0.17 0.010	0.07 0.951	0.02 0.099	1.65 0.016	3.8E-5 0.010
6	0.14 0.008	0.06 0.835	0.02 0.086	1.41 0.013	3.3E-5 0.009
7	0.14 0.008	0.06 0.809	0.02 0.085	1.40 0.013	3.3E-5 0.009
8	0.16 0.009	0.07 0.945	0.02 0.098	1.66 0.016	3.8E-5 0.010
10	0.48 0.026	0.13 2.047	0.05 0.203	2.35 0.022	1.3E-4 0.033
12	0.14 0.008	0.06 0.870	0.02 0.091	1.51 0.014	3.4E-5 0.009
17	0.13 0.008	0.05 0.769	0.02 0.079	1.30 0.012	3.1E-5 0.008
14	0.13 0.008	0.05 0.789	0.02 0.084	1.42 0.013	3.0E-5 0.008
18	0.26 0.015	0.10 1.434	0.03 0.152	2.66 0.025	6.1E-5 0.016
20	0.26 0.015	0.10 1.421	0.03 0.150	2.63 0.025	5.9E-5 0.015
21	0.17 0.010	0.07 0.970	0.02 0.099	1.67 0.016	3.8E-5 0.010
28	0.24 0.014	0.09 1.333	0.03 0.141	2.37 0.022	5.5E-5 0.014
34	0.27 0.015	0.10 1.474	0.04 0.154	2.66 0.025	6.4E-5 0.017
33	0.26 0.015	0.10 1.455	0.03 0.152	2.61 0.025	5.9E-5 0.015
36	0.21 0.012	0.09 1.550	0.04 0.158	2.64 0.025	6.2E-5 0.016
37	0.28 0.016	0.11 1.550	0.04 0.167	2.88 0.027	6.5E-5 0.017

Canister No.	^{238}Pu/ Canister (Ci,g)	^{239}Pu/ Canister (Ci,g)	^{240}Pu/ Canister (Ci,g)	^{241}Pu/ Canister (Ci,g)	^{242}Pu/ Canister (Ci,g)
38	0.17 0.009	0.07 1.102	0.03 0.115	1.72 0.016	4.8E-5 0.013
41	0.28 0.016	0.11 1.580	0.04 0.167	2.95 0.028	6.7E-5 0.017
42	0.27 0.015	0.10 1.455	0.04 0.154	2.77 0.026	6.2E-5 0.016
43	0.26 0.015	0.10 1.484	0.04 0.162	2.76 0.026	6.2E-5 0.016
44	0.26 0.015	0.10 1.460	0.04 0.154	2.64 0.025	6.0E-5 0.016
45	0.25 0.014	0.10 1.387	0.03 0.147	2.52 0.024	5.5E-5 0.014
46	0.26 0.015	0.10 1.382	0.03 0.149	2.62 0.025	5.7E-5 0.015
47	0.24 0.014	0.09 1.289	0.03 0.138	2.30 0.022	5.2E-5 0.014
48	0.22 0.013	0.09 1.245	0.03 0.132	2.27 0.021	5.1E-5 0.013
49	0.19 0.011	0.07 1.075	0.03 0.112	1.89 0.018	4.5E-5 0.012
50	0.20 0.011	0.08 1.089	0.03 0.114	1.99 0.019	4.7E-5 0.012

Table 10 presents the total activity of each plutonium isotope present in the 30 RH-TRU waste canisters summed from Table 9.

Table 10. Total of Plutonium Isotopes in 30 RH-TRU Waste Canisters

Plutonium Isotope Number	Plutonium in 30 RH-TRU Waste Canisters (Ci)	Plutonium in 30 RH-TRU Waste Canisters (g)
238	6.67	.38
239	2.57	37.59
240	.92	3.94
241	65.92	.62
242	1.59E-03	.41
TOTAL	76.08	42.94

PNL analyzed one glass sample for uranium (RH-TRU waste canister 36) and another for americium (RH-TRU waste canister 10). They assumed that the remaining 29 RH-TRU waste canisters contained comparable quantities of americium (based on the relative amounts of plutonium to americium present in Canister 10). Correlation between uranium present in the glass in Canister 36 and other radionuclides could not be made using a single sample; however, PNL assumed that the other canisters contained comparable quantities of uranium. When BWHC performed the necessary calculations to determine the uranium concentrations and activity present in the RH-TRU canisters at the time of shipment, they assumed that all of the canisters contained the same quantity of uranium. Closely related canister weights for the 30 RH-TRU canisters and the absence of a correlation with other radionuclides for uranium support this approach.

Table 11 shows the distribution of the uranium isotopes and content in each canister, and Table 12 shows the isotopes of americium that PNL calculated and provided to BMFT. Appendix 2 contains the letter from PNL to the BMFT (Letter, L. K. Holton to R. Stippler, dated September 22, 1989).

Table 11. Uranium Content in RH-TRU Waste Canister No. 36

Uranium Isotope	Mass Percent	Mass in Canister (g)
234	0.00859	7.7E-4
235	0.87458	7.9E-2
236	0.06831	6.2E-3
238	99.03	8.96

Table 12. Americium Content in RH-TRU Waste Canister 10

Americium Isotope	Quantity in Canister (Ci)	Mass in Canister (g)
241	0.242	0.071
243+244	0.130	No mass data

Each of the 30 RH-TRU waste canisters was electro-polished to reduce smearable contamination to levels below 22,000 disintegrations per minute (dpm) per cm^2 (370 Bq/100 cm^2) beta-gamma and 2,200 dpm per cm^2 (37 Bq/100 cm^2) alpha before it was placed in the water cooled storage unit.[3]

3.2.2 Radiological Characterization of RH-TRU Instrumented Waste Canisters 13 and 19

PNL fabricated RH-TRU waste canisters 13 and 19 during RLFCM Run-10. They calculated the radiochemical content of the two RH-TRU waste canisters using the glass mass in each canister and glass sample analytical results. They calculated the plutonium isotopic concentration based on the quantity of ^{90}Sr in each canister (using the $^{239,240}Pu$: ^{90}Sr ratio of 0.71) (see Appendix 2). The uranium content in the two instrumented RH-TRU waste canisters was assumed to be equal to that of the 30 RH-TRU waste canisters. Using the ratio of americium to plutonium in RH-TRU waste canister 10, where both a plutonium and an americium analyses were conducted, PNL calculated the americium content in the two canisters. Table 13 shows the PNL summary of the radiochemical contents of the two canisters.

Table 13. Radiochemical Contents of RH-TRU Waste Canisters 13 and 19[a]

Radionuclide	Canister 13	Canister 19
^{137}Cs (kCi)	125.9	163.9
^{90}Sr (kCi)	51.6	78.7
^{238}Pu (Ci, g)	3.2E-2 1.9E-3	4.5E-2 2.6E-3
^{239}Pu (Ci, g)	3.1E-3 5.0E-1	4.3E-2 6.9E-1
^{240}Pu (Ci, g)	1.04E-2 4.5E-2	1.4E-2 6.3E-2
^{241}Pu (Ci, g)	4.3E-1 4.3E-3	5.9E-1 5.9E-3
^{242}Pu (Ci, g)	8.5E-6 2.17E-3	1.17E-5 3.0E-3
^{234}U (g)	<8.6E-4	<8.6E-4
^{235}U (g)	<8.7E-2	<8.7E-2
^{236}U (g)	<6.9E-3	<6.9E-3
^{238}U (g)	<10	<10
^{241}Am (Ci)	7.9E-5	1.1E-4
^{243,244}Am (Ci)	4.3E-5	5.9E-5
Decay Heat (watts)	949	1280

[a] Activity in plutonium isotopes calculated using 49 CFR 173.435 Table specific activity in Ci/g.

4.0 CHEMICAL CHARACTERIZATION

PNL analyzed all chemical constituents in the RH-TRU waste canisters through inductively coupled plasma (ICP) emission spectroscopy analysis.[3] They determined the weighted average composition of each RH-TRU waste canister by using individual glass sample analyses and then weighting the analysis results by the mass of the pour from which each sample was obtained.[3]

Table 14 contains the average glass composition for the 30 RH-TRU waste canisters PNL fabricated under the BMFT original agreement, the two instrumented RH-TRU waste canisters (numbers 13 and 19), and the two test RH-TRU waste canisters (FRG A and B). The average glass composition for the 30 RH-TRU waste canisters is shown by RLFCM Run number 7, 8, or 9.

Table 14. Average Glass Composition of the FRG Canisters

Oxide Compound	Average Glass Composition, wt% RLFCM-7[a]	Average Glass Composition, wt% RLFCM-8[a]	Average Glass Composition, wt% RLFCM-9[a]	Average Glass Composition, wt% RH-TRU Waste Canister A/B[b]	Average Glass Composition, wt% RH-TRU Waste Canister 13/19[c]
Al_2O_3	2.88	2.58	2.17	2.18	2.60 / 2.34
B_2O_3	13.68	14.65	14.84	16.43	15.10 /15.06
BaO	1.05	1.13	1.02	0.12	1.06 /1.04
CaO	1.52	1.25	0.79	0.64	0.57
CeO_2	0.06	0.05	0.07	0.17	0.24 /0.10
Cr_2O_3	0.58	0.38	0.45	0.24	0.68 /0.70
Cs_2O	5.02	2.08	5.74	0.11	3.88 /4.95
Fe_2O_3	11.18	10.10	9.93	12.42	10.91 /10.39
La_2O_3	1.04	1.07	1.53	0.09	1.60 /1.46
Li_2O	0.31	0.00	0.00	0.21	--
MgO	0.78	0.54	0.44	0.17	0.48 /0.46
MnO_2	0.80	1.20	1.11	1.36	0.99 /0.92
MoO_3	0.05	0.00	0.00	--	--
Na_2O	16.50	13.22	11.58	15.58	12.51 /12.68
Nd_2O_3	0.65	0.71	0.89	0.18	1.10 /1.08
NiO	0.39	0.25	0.44	0.29	0.41 /0.32
PbO	0.16	0.00	0.00	0.00	--
RuO_2	0.02	0.00	0.00	0.12	--

Oxide Compound	Average Glass Composition, wt% RLFCM-7[a]	Average Glass Composition, wt% RLFCM-8[a]	Average Glass Composition, wt% RLFCM-9[a]	Average Glass Composition, wt% RH-TRU Waste Canister A/B[b]	Average Glass Composition, wt% RH-TRU Waste Canister 13/19[c]
SiO_2	41.25	48.02	46.59	47.07	46.70 /46.41
SrO	1.65	2.67	2.34	0.04	1.10 /1.42
TiO_2	0.19	0.07	0.03	0.64	0.03 / --
ZnO	0.08	0.01	0.00	0.00	0.01 / --
ZrO_2	0.15	0.04	0.05	0.18	--/ --
Rare Earth Mix	--	--	--	0.13	
TOTAL	100.00	100.00	100.00	99.7	99.97 /99.90

(a) From PNL-6790, Processing Summary Report: Fabrication of Cesium and Strontium Heat and Radiation Sources, February 1989

(b) From information provided by Y. B. Katayama, BWHC on 11/12/96.

(c) From 325 ICP Analyses of Canister 13 and 19 glass analyses.

5.0 WASTE DESIGNATION

The B&W Hanford Company (BWHC) Project C-229 Manager designated the 34 RH-TRU waste canisters as non-hazardous, remote-handled transuranic waste, based upon an earlier PNL generator waste designation study (see Appendix 3).

The source material for the radionuclides in the FRG A and B canisters is spent fuel discharged from the Point Beach reactor in Two Rivers, WI in May 1974[6] (burnup was 29600 Mwd/tU). This fuel was obtained by DOE for use in the Nuclear Waste Vitrification Project, the objective of which was to demonstrate the vitrification of actual high-level liquid waste from spent LWR fuel.

The spent fuel underwent chop/leach and dissolution in the 324 Building and was then sent to the nearby 325 Building for separation. Following separation, the aqueous fission product stream was returned to the 324 Building and incorporated into a borosilicate glass to demonstrate the vitrification process. Residual fission product stream material from the NWVP was subsequently used to produce the radioactive glass needed to check out and demonstrate the new Radioactive Liquid Feed Ceramic Melter (RLFCM) and the rest of the equipment designed for the FRG project.[7] This "spiked" glass was necessary to evaluate melter instrumentation such as liquid level sensors . Thus, this glass was "product" insofar as it was required to develop the RLFCM and confirm its proper operation and confirm the capability to fabricate and characterize the FRG canisters. Following checkout of the melter, the

glass was poured into canisters A and B and further tests were performed to verify the FRG canister welding process and the canister characterization devices. Thus, the canisters themselves were "product".

The source material for the radionuclides in the other 32 FRG canisters came from B-Plant as described previously (see Section 1.2.1). Both the Cs and Sr became "product" after being high-level liquid waste originating from the separation of defense-related fuel, when they were diverted to fabrication of the FRG isotopic heat sources. Only after the project was cancelled by the Federal Republic of Germany did these canisters, and the project test canisters, become waste.

All canisters possess greater than 100 nCi/g alpha-emitting transuranic radionuclides with half-lives exceeding 20 years. Canister FRG A has the least plutonium content (the more abundant transuranic element in this waste) with 0.0207 g (see Section 3.1). Including the transuranic radionuclides ^{237}Np and ^{241}Am but discounting the ^{241}Pu content (because it is not an alpha emitter) results in 334 nCi/g (including the weight of the metal canister with the glass). Canister FRG B contains 0.0579 g Pu; the remainder of the 32 canisters contain at least ten times that quantity. Thus, canister FRG A bounds all of the others and all exceed 100 nCi/g.

As disposition of the canisters was being considered, potential leachability of hazardous constituents was investigated. The December 1993 PNL study[9] discussed leachability tests[10,11] conducted on various forms of borosilicate and aluminum oxide based glasses that did not contain radionuclides. Within the study was a brief discussion of toxic characteristic leachate procedure (TCLP) studies conducted on two batches of non-radioactive glass that were similar in composition to the glass used for the 30 RH-TRU canisters. The concentrations of lead and chromium in leachate from the two batches was said to be 0.000 to 0.043 parts/million (ppm) for lead and 0.005 to 0.013 ppm for chromium, well below the concentrations regulated in WAC 173-303-090(8)(c).

BWHC personnel also investigated the research that supported the 1993 PNL waste designation study and more recent evaluations of glasses being considered for waste immobilization. They found that actual values for the non-radioactive surrogate glass resulting from the TCLP evaluation performed by PNL[11] showed no values in excess of regulatory limits: The maximum concentration of barium in the leachate was 0.1179 ppm (limit in leachate is 100 ppm).[11] The concentration of lead was 0.0174 ppm (limit in leachate is 5 ppm), and the concentration of chromium was 0.0096 (limit in leachate of 5 ppm).

When BWHC reviewed recent technical evaluations[12,13] of glass to be used for waste immobilization, they found that the evaluations showed that the impurities and constituents of the glasses are not preferentially leached. Concentrations in the leachate are approximately proportional to their concentrations in the glass (i.e. their fractional dissolution values). Elements for which no TCLP analysis appears (the presumption being that none was detected) should not be present in greater concentration than the ratio of the TCLP concentration of the analyzed element to its concentration in glass multiplied by the concentration of the non-analyzed element in the glass. BWHC used that assumption to evaluate the surrogate glass and determined the range of values for fractional dissolution. From the analysis they performed, BWHC was able to confirm that the metals in the glass would not appear in concentrations regulated by the Washington Dangerous Waste Regulations. The

BWHC analysis and supporting documentation will be provided as part of the Project C-229 Waste Portfolio.

6.0 SEGREGATION

The 34 RH-TRU canisters will be placed in the casks described in PACKAGING below. No other radioactive or mixed wastes will be packaged with the canisters.

7.0 PACKAGING

The canisters containing the RH-TRU waste sources will be transported from the A-Cell in the 324 Building to the Interim Storage Area within the Central Waste Complex in the 200 West Area via truck or rail. Two types of packagings will be used, with different materials of construction (steel, nodularized iron) and capacities (three canisters, five canisters. Each type of packaging is covered by a Safety Analysis Report for Packaging.[14,15]

7.1 GNS-12

The GNS-12 cask was fabricated in Germany and certified as a Type B(U) packaging that complied with the International Atomic Energy Agency (IAEA) Safety Series No. 6 requirements for the transport of the canisters containing the isotopic heat sources. The unmodified packaging held a Certificate of Competent Authority [USA/0441/B(U), Rev. 0 (1991-1993)] from the U.S. Department of Transportation (DOT) for the same material. The packaging was modified for storage on the Hanford Site to incorporate metallic seals in the inner seal ring of the cask lid.

The GNS-12 packaging was originally designed to transport the isotopic heat sources (^{137}Cs and ^{90}Sr borosilicate glass) encapsulated in stainless steel canisters from the U.S. to the Federal Republic of Germany. Designed to hold up to three canisters, the packaging is constructed as a composite container of outer and inner stainless steel shells with lead shielding between the shells.

The GNS-12 cask body is constructed of austenitic stainless steel (U.S. equivalent to Type 304) sheets welded into cylindrical inner and outer shells, bottom shells, and stainless steel parts to accept a lid. The void between the inner and outer shells is filled with lead. The inner shell is 2.0 cm thick, the outer shell is 4.0 cm thick, and the lead inner layer is 10.0 cm thick (16 cm total). The cask cavity is 72.3 cm in diameter and 122.0 cm tall. When unloaded, the cask body weighs 8600 kg.

On the outside wall of the cask body, longitudinal fins were welded in the region of the cask cavity to aid in removal of thermal heat. No active heat removal system was incorporated into the cask.

Two diametrically opposed lifting trunnions are bolted to a corresponding seat welded to the upper lateral surface of the cask body. They allow the cask to be lifted by the trunnions using slings or cables.

The cask body, cask lid, and seal system comprise the containment boundary. The top of the cask was constructed to allow the lid to be secured to the cask body. The lid is a disk of stainless steel 14.0 cm thick that is bolted to the cask body using 20 cap screws, size M30. Bolted to the bottom of the cask

lid is an 8.0-cm thick lead plate that extends into the cask cavity space to improve radiological shielding. Three positioning pins at three intermediate locations facilitate cask lid centering during loading and unloading. The total weight of the lid is 1055 kg.

The cask lid sealing surface contains two machined concentric seal (O-ring) grooves for a combination metallic/elastomeric seal. The seal system includes two seals. The inner seal is a metallic seal that meets the requirements for 20 years of interim storage. It is an Inconel helically-wound spring encased in a stainless steel C-shaped section inner jacket and an outer C-shaped aluminum section outer jacket. The outer seal is a larger diameter elastomeric O-ring.

The cask lid contains several penetrations. Two penetrations were designed to perform necessary operations during loading and unloading. 1) The water flooding and drainage port will not be used because the canisters will be loaded into the casks in the hot, dry environment of the 324 Building A-Cell. 2) The other port allows for evacuation and/or backfilling the cask with helium. It is closed with a quick disconnect coupling. Both penetrations are covered with seal cover plates with one elastomeric and one metallic seal that can be inspected via a test port.

Other penetrations allow the performance of leak tests on the storage, transportation, and port cover seals. One test port allows access to the space between the cask lid seals for connection of a leak test adapter. Another test port located on the cask lid outboard of the elastomeric seal provides an annulus for leakage rate testing of the elastomeric transportation seal. One tapered test port, 2.5 cm in diameter tapered to 1.0 cm, and two 14.8 cm-diameter countersunk ports are machined into the top side of the cask lid. The test port that penetrates the void space between the transportation seal and the storage fitting after testing is 12.8 cm long and 1.8 cm in diameter. This test port is fitted with a quick-disconnect fitting to allow helium backfilling/detecting processes required for performance of cask cavity leak tests. The test port is sealed after testing with a metallic seal ring. The seal cover ports have two machined grooves to accommodate metallic/elastomeric seals. The metallic/elastomeric seals are compressed to the proper specification by the installation of the seal cover ports.

Inside of the cask body is a solid aluminum alloy (type Al Mg 4.5 Mn) basket structure that is used to immobilize the canisters during transport. The basket was designed to allow heat transfer from up to three canisters. Three thru-holes, each with an inside diameter of 31.0 cm, serve as insertion guides and support members. The thru-holes are placed on a 37.5-cm pitch circle. Lifting eyes on the basket allow it to be lifted. The basket weighs 620 kg and is 103.0 cm long.

During transport, the GNS-12 is fitted with upper and lower impact limiters constructed of sheet metal filled with wood. Each impact limiter weighs 715 kg. The impact limiters serve as impact absorbers and thermal insulators during the normal conditions of transport and during accidents. The impact limiters are attached to each other by tension rods and bolts that are in turn attached to the bottom of the transport container. The top impact limiter is 165.0 cm outside diameter by 41.5 cm long; the bottom impact limiter is 165.0 cm outside diameter by 38.5 cm long.

The loaded GNS12 cask weighs 11075 kg without the impact limiters, based on an assumed weight of 250 kg per canister. When the impact limiters are added to the packaging, the total package (i.e., cask, canisters, impact limiters) weighs 12500 kg maximum.

During transport, the GNS-12 casks are transported in a closed transport container. The Open All System opens by a sliding hinge system (roof/three sides) that allows full access to the casks for loading and unloading. One side is stationary and contains an access door, name plates, and total package weights. The access door allows entry after loading and securing for inspection and monitoring during transport or storage. The casks are attached to a supporting frame as part of the tie-down system in the Open All System. When two casks are loaded into the Open All closed transport container, the total weight of the container (two casks, impact limiters, and transport container) will be 29000 kg (64,000 lb).

7.2 CASTOR GSF

The CASTOR GSF cask was designed and fabricated in Germany to be a certified Type B(U) packaging that complied with the IAEA requirements for transport. The cask was never actually certified by the IAEA for use.

The CASTOR GSF cask assembly is comprised of a ductile cast iron (type GGG40) cask body, a GGG40 primary lid that is leak testable, a structural mild steel (carbon steel 52.3) secondary lid, metallic seals and elastomeric seals for the primary and secondary lids, an A-36 carbon steel and aluminum basket to immobilize the canisters, and impact limiters (glued plywood boxed in steel 37.2) for each end. The cask is painted with a decontaminable paint to facilitate decontamination and to shed precipitation.

The cask body and primary lid system form the containment for the package. The main body of the cask is a hollow ductile cast iron (GGG40) cylinder. The GGG40 outside wall is 220.0 mm thick, and the bottom outside wall is 205.0 mm thick. The steel inner liner is 2.0 mm thick. Between the inner and outer walls of the cask is a lead layer that is 28.0 mm thick on the sides and 55.0 mm thick on the bottom. The cask inner cavity is 1250.0 mm high, and the inner diameter is 895.0 mm. Without the impact limiters, the cask is 1795 mm tall; with the impact limiters, the cask is 2475.0 mm tall.

The cask body was constructed with cooling fins on its surface to improve heat removal; no active heat removal system was incorporated into the casks. The fin region is 1300.0 mm long. Each fin is 140.0 mm tall, tapering from 20.0 mm at the base to 10.0 mm at the head. From the tips of the fins, the cask is 1675.0 mm in diameter.

The primary lid is a disk-shaped cylindrical structure constructed of 180-mm thick GGG40. Where the primary lid meets the cask surface, two grooves are machined to accept an inner metallic seal and an outer elastomeric seal. The lid is reduced in diameter at the bottom and extends into the cask; it has a recess in the center of the top side to permit remote handling. A 55-mm thick lead plate bolted to the bottom of the primary lid shields against gamma radiation.

The primary lid is bolted to the cask body by 24, cylinder-head, high strength carbon steel bolts, size M36. To center the lid during loading and unloading, three of the bolts are replaced by guide pins. Three tapped blind holes

around the center of the lid allow attachment of the lifting gear during loading and unloading. Without the bolts, the lid weighs 1445 kg.

The primary lid contains three penetrations to aid handling operations during loading and unloading. One port used for evacuation or charging the interior with helium is closed with a quick-disconnect coupling and a metal seal ring. The port is covered by a protective cap provided with an inner metal seal and an outer elastomeric seal. The second test port allows access to the space between the seals of the primary lid for connection of a leak test adapter. The second port is closed with a screw plug sealed with an elastomeric ring. The third test port can be used for monitoring the quick disconnect coupler during storage; it is sealed with a screw plug with a elastomer seal.

The secondary lid is also a disk-shaped cylindrical structure with two grooves machined on the inner surface to accept an inner metallic seal and an outer elastomeric seal. Constructed of structural mild steel (carbon steel 52.3), the secondary lid attaches to the cask body with 24, high-density carbon steel, cylinder-head bolts, size M24. The secondary lids forms a second seal barrier and permits pressure monitoring between the lids during storage.

The secondary lid has four penetrations. One port, used for evacuation or charging the barrier space with helium during storage, is closed with a quick-disconnect coupling and a metallic sealing ring. That port is covered by a seal cover with an inner metal seal ring and an outer elastomeric seal ring. The second port is allows access to the space between the secondary lid seals for connection of the leak test adapter. The second port is sealed with a screw plug sealed with an elastomeric ring seal. A third port is designed to accept a pressure switch. During transport, a cover is placed over the port that is sealed with an elastomeric seal ring. A fourth port used to monitor the quick disconnect coupling during storage is sealed with a screw plug with an elastomeric seal ring.

The canister basket is a carbon steel tubular structure with five positions in a pitched circle configuration that was designed to hold up to five canisters. Individual position tubes are formed by a continuous steel tube (inside diameter 312.7 mm; wall thickness, 5.6 mm); the tubes are surrounded by a cylindrical sheet-metal shell (6.5 mm thick). Between the position tubes and sheet-metal shell is cast lead to improve heat removal and shielding. A lifting eye in the center of the basket allows handling.

Impact limiters on the head and foot of the CASTOR GSF cask are connected together by tie rods to reduce shock loading during an accident in transit. The impact limiters are constructed of closed sheet-steel filled with several plies of pine. They extend from the end faces of the cask to the beginning of the fin region to completely enclose the cask head and foot. Two lifting eyes attached to the top surface of the top impact limiter in diametric opposition allow lifting. Two lifting eyes attached to the bottom impact limiter aid placement on the shipping pallet.

A special 20-ton lifting beam provides for cask handling via two lifting rings in diametrical opposition on the outside of the cask body. During transport, the loaded CASTOR GSF packages will be secured to a rail flat car or truck bed.

When the CASTOR GSF package is assembled (cask, canisters, impact limiters, skid) for shipping, it weighs 23930 kg.

Figures 1 through 6 show the GNS 12 and the CASTOR GSF casks and lids.

7.3 Packaging at 324 Building

The canisters will be retrieved from their storage locations within Cell A and placed in the casks in accordance with approved procedures. Draft procedures now in review within BWHC will be approved and issued before the canisters are loaded on the casks. The loaded casks will be sealed and leak-checked, as required by the Safety Analysis Reports for Packaging (HNF-SD-TP-SARP-021 and -022).

7.4 Gas Generation

The GNS-12 and CASTOR GSF SARPs[14,15] evaluated gas generation within the loaded casks and determined that there is no gas generation caused by the 30 RH-TRU waste canisters and instrumented RH-TRU canister 13 within the GNS-12 or CASTOR GSF packagings. The solid borosilicate glass matrix is encapsulated within stainless steel canisters that were welded then leak tested to 1.0E-7 mbar/s, air (1.0E-7 atm · cc/s, air). The welded canisters prevent the release of any gases from the borosilicate glass into the inner cavity of the cask. If the stainless steel canister disintegrates during an accident leaving only the borosilicate glass, the gases released will be negligible.

The three remaining canisters do not meet the leak rate criteria. The two RH-TRU test canisters (FRG A and B) were not leak tested because they were not fabricated to meet BMFT standards. Instrumented RH-TRU canister 19 underwent the gross leak check of its lid weld; however, it failed the sensitive leak check. When the three canisters are loaded into a cask, the cask will be leak checked to ensure that the cask cavity is sealed.

7.5 External Shielding

The 34 RH-TRU waste canisters require shielding from gamma emitters only (^{137}Cs, $^{90}Sr/^{90}Y$). During transportation (HRCQ exclusive use), the accessible surface radiological dose rate for the package must be less than 2 mSv/h (200 mrem/h), less than 0.1 mSv/h (10 mrem/h) at 2.0 m from the package, and less than 0.05 mSv/h (5.0 mrem/h) in the cab of the transporter, per the approved SARPs.[3,4] When the casks are in storage, the accessible surface must be less than 2 mSv/hr (200 mrem/h). Direct dose measurements and representative smear surveys will be collected to ensure that the SARP Section A4-1 transport criteria are met before the casks are shipped.

Analyses compiled in Table B5-4 in each of the SARPs showed that the dose rate at any point on the surface of the casks did not exceed 100 mrem/hour. The surface dose will not therefore exceed the limit of 100 mrem/h for transuranic waste in the Hanford Site Solid Waste Acceptance Criteria (WHC-EP-00163 Rev. 4).

7.6 Heat Dissipation

The maximum thermal capacity of a single GNS-12 cask is 6855 watts, or 2285 watts per canister. The maximum temperature at the exterior of the two GNS-12 casks inside of the Open All container is less than 82° C, as determined in SARP Sec. B8.2.[15]

The maximum thermal capacity of a single CASTOR GSF is 11425 watts, or 2285 watts per canister. The maximum temperature at the exterior of the loaded CASTOR GSF cask < 85° C, as determined in the SARP Section B.1.2.7.[14]

8.0 WASTE MINIMIZATION

To minimize waste, the 34 RH-TRU canisters will be separated from excess equipment in the A Cell, and packaged and stored separately.

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